Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/713
FINLAND
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

On Nonlinear Reactor Kinetics

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The nonlinear features of reactor kinetics including the effect of the prompt negative temperature coefficient of reactivity are studied both experimentally and theoretically. Two extreme cases are considered: transients produced by an instantaneous addition of reactivity and the steady state response of the reactor to sinusoidal variation of reactivity. The experiments have been performed with a TRIGA Mark II reactor.

The unsymmetry of the produced pulse and its succeeding "tail" are studied in the transient experiments. The experimental determination of the pulse is carried out by three different methods: 1) by using a boron coated ionization chamber in connection with a recording oscillograph, 2) by recording the intensity of the Čerenkov radiation in the core with a photomultiplier fitted to an appropriate optical system, and 3) by using a multichannel time analyzer to record the intensity of neutron capture gamma rays from the 8.2 eV resonance of samarium. The subsequently developed theory shows that the measured properties can be described satisfactorily by the properties of the delayed neutron groups and the transfer of heat from the fuel to the water moderator.

The response of the reactor to a sinusoidal variation of reactivity is studied in terms of the fundamental and some lower order harmonic as well as subharmonic components. The analysis of the response is performed on the basis of physical concepts as well. The nonlinear transfer functions of the reactor for reactivity variations of larger amplitudes are derived and their relationship to the transient properties are discussed. Finally, the stability of the reactor under normal operating conditions is studied and is found to be satisfactory.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/714
WMO
February 1964
Submitted in:
ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

RECENT DEVELOPMENTS IN THE APPLICATIONS OF METEOROLOGY TO REACTORY SAFETY

by

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Recent advances in nuclear technology have required, and produced, new meteorological tools and techniques which contribute to more refined evaluations of reactor site suitability, more precise estimations of the atmospheric dispersion, and new techniques of measuring air flow.

There is a growing tendency to design meteorological programmes to match the scope and particular problems of specific nuclear installations.

Among the new approaches to reactor siting which are described are nation-wide analyses of vertical lapse rate and methods for determining the persistence of wind regimes. The use of conventional meteorological observations to infer diffusion regimes is described and illustrated for several locations.

Use of wind fluctuations to obtain diffusion coefficients has been improved and empirical techniques developed that permit use of conventional wind systems.

Smoke photography techniques that permit quantitative analysis are described, together with illustrations of the field use of particulate tracer techniques in special problems.

One of the newest meteorological techniques for application to nuclear sites is the use of balloon borne radar beacons to obtain air trajectories and diffusion data over ranges up to 100 miles. This technique, and its potential, is described in considerable detail.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/715
ARGENTINA
February 1964
Original: SPANISH

Confidential until official release during Conference

Abstract

THE CONTRIBUTION OF NUCLEAR ENERGY TO THE ARGENTINE ENERGETIC PROBLEM

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This paper outlines the general characteristics of the Argentine Republic and its present energetic situation. The energy resources, power availability and electric power demand, both present and estimated for the future, are specially discussed. A brief outline of the country's nuclear stand is offered.

After stating the Argentine energetic problem, possible alternative solutions are discussed, considering their respective advantages and disadvantages. The possible contribution of nuclear energy is dealt with in further detail.

The financing aspects are specially analysed, because of their fundamental importance for a developing country.

The paper ends with a discussion of all the factors and criteria utilized in selecting the type, power and location of the first nuclear power station to be installed in Argentina.

Translation provided by courtesy of the submitting Government

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/716
BELGIUM
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

FUEL ELEMENTS IN THE FORM OF CONCENTRIC CYLINDRICAL PLATES

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Research reactor fuel elements, containing enriched uranium, in the form of concentric cylindrical plates or tubes and cooled by water, as in the BR2 reactor at Mol, offer a wide range of possibilities for their adaptation to enable them to satisfy the diverse requirements for in-pile irradiation facilities.

This paper examines this type of fuel element from the point of view of the problems of manufacture, the test and commissioning programme of the fuel element and gives a description of the instrumentation used on in-pile tests on the different variations of the basic element; a description of the fuel element developed for the ESSOR (Euratom) reactor is given.

Details of the fabrication of this type of fuel element have already been described. Strict control is necessary during manufacture to guarantee the best possible distribution and homogeneity of the U^{235} and a complete bond between the fuel and its cladding.

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A fuel element similar to the BR2 one is proposed for the ESSOR reactor. The characteristics are given and the fabrication procedure is described.

Prototype non-fissile models of the fuel elements were submitted tc hydraulic flow tests in a water loop; temperature gradients were not created in the plates. These particular studies were concentrated on water velocity measurements; a particular point of interest is the hydraulic gauge specially developed for this work.

Results are given for the group of prototypes tested; a description of the test set-up is given in an appendix. Neutron measurements are made with the actual elements in the zero energy model of the reactor. These measurements enable the determination of a series of characteristics for a group of elements or for a particular element and to obtain the fine flux distribution inside the element.

To obtain sufficient information concerning the performance of the fuel elements in-pile a special instrumentation was developed; temperature probes to obtain the heat balance in a reactor channel occupied by a fuel element and thermocouples fixed to the plates of the elements to verify the temperatures obtained by calculation. Description is given of the other tests which show the influence of certain operational parameters.

The fuel elements have been operated at a thermal flux of 600 watts/cm^2 and the paper concludes with an attempt to determine the limiting factors of this type of element.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/717
IAEA
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

SURVEY OF NUCLEAR DATA FOR REACTOR CALCULATIONS

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Since the 1958 Geneva Conference, developments in measuring methods have resulted in a great increase in the amount of detail available for neutron cross-section data for important nuclei as a function of the energy of the incident neutron. Improvements in techniques, especially for fast neutron studies, and in resolution in the "resonance" region, have been most marked. In spite of this, and an accompanying increase of accuracy of the data, a number of important uncertainties and inconsistencies remain, which will be reviewed in the paper. The extent to which further increases of detail or accuracy are still needed will also be briefly considered. A few quantities will be selected for attention to illustrate the problem, in particular the thermal neutron constants for the fissile nuclei of U and Pu.

The compilation and evaluation of nuclear data and its preparation for use in particular reactor computation codes has become a quite sophisticated process and the part which the IAEA can play in this work will be briefly discussed. The sources of information in this field and the published volumes of compilations will be reviewed, from the point of view both of the effort involved in their production and of their use for particular applications. Particular difficulties concern the need for critical judgement in reconciling discordant measurements and interpreting certain effects connected with the experimental techniques used. There are also cases where no measurements exist and where theory has to be used to provide values. Finally, the result is a set of values of the cross-section or other quantity involved, as a function of neutron energy; from this group averages or other needed values can be derived.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/718
BYELORUSSIAN SSR
February 1964
Original: RUSSIAN

Confidential until official release during Conference

Abstract

PHYSICAL CHARACTERISTICS OF THE POOL-TYPE RESEARCH REACTOR IRT-2000 WITH TEST LOOP

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The setting of an experimental loop channel inside the core of a research reactor with a high specific power and intensive thermal neutron flux is of interest for performing various experimental works requiring intensive neutron and γ-fluxes as the test conditions.

In the very core of the reactor IRT-2000 with a power of 1700 kW thermal neutron fluxes attain the magnitude of $3 \cdot 10^{13}$ neut/cm^{2}s neutron fluxes with energy over 0.2 eV., the magnitude of $6 \cdot 10^{13}$ neut/cm^{2}s and energy fluxes of γ-radiation, the magnitude of $1 \cdot 10^{14}$ MeV/cm^{2}s, and the rate of γ-dose becomes $1 \cdot 10^{8}$ rad/h.

Such a relatively high degree of radiation allows the experiments on irradiation and on testing checking the radiation resistance of materials as well as various loop experiments to be carried out. Physical characteristics of the water-moderated, water-cooled reactor IRT-2000 with a loop channel in the centre of the core were investigated on the electronic digital computer "Ural-1" of the Institute of Mathematics and Computing Engineering of the BSSR Academy of Sciences.

It is found that the internal reflector around the testing loop channel allows to considerably intensify the thermal neutron flux in the channel region and thus to increase experimental possibilities of IRT-2000.

Beryllium oxide, graphite, water and operating cassettes are considered as the material of the internal reflector. The data are given on the critical mass, reactivity storage, as well as on the fluxes of thermal-moderated and fast neutrons for the mentioned materials of the internal reflector and a number of compositions of the core.

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The calculated minimum critical mass of the reactor without a channel with a graphite external reflector, equal to 2520gr.U^{235}, differs from the measured one by 2 per cent.

The minimum critical mass of the reactor with a loop channel, containing more than 3.3 kg. of stainless steel and equal to 3800 gr. of U^{235} is attained in the presence of BeO as an internal reflector.

The maximum reactivity storage and maximum reactor life with a channel are to be observed in the case when operating cassettes are used as an internal reflector.

It is shown in the work that the critical masses of a water-moderated, water-cooled reactor with a loop channel in the centre of the core can be satisfactorily described in the limit of 3-group method, using only the data on neutron microscopic cross-sections and fission neutron age in water.

Multigroup constants are obtained according to the effects of self-shielding of U^{235} and U^{238} resonances and neutron leakage through the gas channel in the core and reflector.

Considerable influence of these effects on the critical mass of a thermal reactor is revealed.

The calculation results on the critical masses and neutron-flux distributions in the core with a channel were compared with the experimental data. This comparison showed a good agreement between the theoretical and experimental results.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/719
BYELORUSSIAN SSR
February 1964
Original: RUSSIAN

Confidential until official release during Conference

Abstract

THE MODELLING METHOD ON TRANSIENT THERMAL PROCESSES IN GAS-COOLED REACTORS ON THE ANALOGUE COMPUTERS

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The analogue computers are used more often for the transient investigation with power change, heating, starting-up and reactor shut-down cooling. The use of these computers deteriorated as the transient heat transfer in reactor is described by partial differential equations.

Digital computers permit partial equations to be solved. The difficulty of direct connections of these computers with the real control system elements as well as complicated and laborious programming and programme arrangement is a considerable obstacle to their application for engineering calculations.

The preparation of equation solution being simple, it is not difficult for the analogue computers to connect with the real elements of control systems.

However serial analogue computers (MN-7, EMU-8, EMU-10, MN-8, MN-10, MN-14, MN-6) are designed to solve usual differential equations. Therefore the method for transformation of partial equations into ordinary non-linear differential equations is necessary.

This work offers an effective and simple transformation method of partial equations into ordinary differential ones which take into account parameters distributions along the reactor length by the presentation of the average-integral on the length gas temperature in terms of inlet and outlet temperatures and their derivatives in the form of a series.

The method is checked up at deep separate and simultaneous discontinuous and sinusoidal disturbances over the gas-flow-rate reactor neutron power and inlet gas temperature.

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Two terms of above series provide good accuracy of the solution (error is not more than 3-4 per cent).

The method proposed is compared with finite difference method, taken as a reference one and it is shown that this method is more economic than the finite difference one (the reactor is described by 5 differential equations rather than by 8 using the finite difference method).

The method may be effectively used for the dynamic characteristic study of a reactor, refrigerator and other heat exchangers of nuclear power plants.

This method may be used for the dynamic characteristic study of a gas-cooled reactor with energy change, heating-up, starting-up and reactor shut-down cooling.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/720
BYELORUSSIAN SSR
February 1964
Original: RUSSIAN

Confidential until official release during Conference

Abstract

SLOW NEUTRON SCATTERING BY THE BONDED HYDROGEN

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This paper is mainly based on the Van Hove's original work on the introduction of space time correlation functions. Inelastic scattering of slow neutron is considered on hydrogenous substances since the prevalence of neutron incoherent scattering by hydrogen considerably simplifies the interpretation of the theoretical and experimental results. This choice is caused as well by the importance of hydrogenous substances for the reactor engineering.

The state of the general theory of scattering function on the contemporary stage of the thermalization theories considered and the perspective of using the different approximation of scattering function is discussed in this paper. Considered is the relation between the Van Hove's space time correlation functions and the temperature-dependent Green's functions which is the powerful apparatus for solving the number of problems on the statistical physics.

Since the spherical harmonic method is most commonly used calculation method of the space energy distribution of neutrons in a reactor the angular moments of differential cross section of the slow neutron scattering are calculated in this paper.

The calculations are mainly carried out for water and zirconium hydride on electronic digital computers "Minsk-2".

To compare their efficiency various approximations models of differential cross sections of slow neutron scattering are used for calculations.

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The calculation data obtained are compared with the available experimental data. The calculation results on $\sigma(E)$, $\int\sigma(E)$, $M(E)$ are given in diagrams and tables.